Dated: August 10, 1995.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 95–20236 Filed 8–15–95; 8:45 am]

BILLING CODE 7590–01–M

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from July 21, 1995, through August 4, 1995. The last biweekly notice was published on Wednesday, August 2, 1995 (60 FR 39430).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By September 15, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: March 15, 1995, as supplemented on June 29, 1995.

Description of amendments request: The proposed amendments would revise the Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2, Technical Specifications (TSs) Section 6, ''Administrative Controls,'' to be consistent with the guidance provided in NUREG-1432, ''Standard Technical Specifications, Combustion Engineering Plants.'' The proposed changes will relocate several requirements to other documents and programs consistent with NUREG-1432 and other NRC guidance addressing the administrative section of the TSs such as the ''Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors,'' published in the Federal Register on July 22, 1993 (58 FR 39132).

The Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TSs to licensee-controlled documents, consistent with the standard enunciated in Portland General Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that ''technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.'' The policy statement encouraged licensees to adopt the applicable improved STSs and provided some guidance for the conversion from the present plant-specific TSs to the improved Standard TSs.

The proposed changes will provide significant human factors improvement to the TSs by accomplishing the following: (1) relocating existing requirements to licensee controlled documents consistent with the policy statement; (2) eliminating requirements which duplicate regulations; (3) relocating similar requirements within the same section; (4) editorial changes; and (5) adding requirements consistent with NUREG-1432.

In addition, the licensee proposes dual rolls for the Shift Technical Advisor (STA) and the establishment of a TS Bases Control Program. Allowing the STA to perform dual rolls is not permitted by the current TSs, but the current NRC guidance allows the STA to perform a dual roll. The proposed new TS Bases Control Program will define the appropriate methods and reviews required to implement a TS Bases change which is also consistent with the current NRC guidance. Two other proposed changes, not specifically covered by the above groupings, include a reduction in reporting requirements and utilizing a more effective option for estimating doses.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Relocating existing requirements to Baltimore Gas and Electric Company (BGE)-controlled documents, eliminating requirements which duplicate regulations, locating similar requirements within the same sections and making necessary editorial corrections to incorporate the proposed changes provide Technical Specifications which are easier to use. Because existing requirements are relocated to established BGE programs where changes to those programs are controlled by regulatory requirements, there is no reduction in commitment and adequate control is still maintained. Likewise, the elimination of requirements which duplicate regulations enhances the usability of the Technical Specifications without reducing commitments. Locating similar requirements within the same sections and making necessary editorial corrections to incorporate the proposed changes neither add nor delete requirements, but merely clarify and improve the readability and understanding of the Technical Specifications. Since the requirements remain the same, these changes only affect the method of presentation and would not affect possible initiating events for accidents previously evaluated or any system functional requirement. Therefore, the proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Since the Shift Technical Adviser (STA) is not considered an initiator to any previously evaluated accident nor considered in the accident's response, the use of a dual role STA would not increase the probability or consequences of any previously evaluated accident.

The Technical Specification Bases Control Program provides controls which ensure appropriate reviews of changes to the Bases. Because NRC approval is still needed for changes to the Bases which affect the Technical Specifications, the proposed Program would not affect the probability or consequences of a previously evaluated accident.

Eliminating the requirement for submitting two reports which place unwarranted administrative burden on both Baltimore Gas and Electric Company and the NRC has no affect on the probability or consequences of an accident previously evaluated. Therefore, the proposed changes would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Replacing the film badge with the electronic personal dosimeter provides a more effective, efficient, state-of-the art option for estimating dose and would not impact accidents previously evaluated. Therefore, the proposed change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different type of accident from any accident previously evaluated.

As discussed previously, relocating existing requirements to BGE-controlled documents, eliminating requirements which duplicate regulations, locating similar requirements within the same sections and making necessary editorial corrections to incorporate the proposed changes will not affect any plant system or structure, nor will it affect any system functional or operability requirements. Consequently, no new failure modes are introduced as a result of the proposed changes. Therefore, these types of changes would not create the possibility of a new or different type of accident from any accident previously evaluated.

Because the STA does not perform equipment design or equipment manipulation, the use of a dual role STA would not create the possibility of a new or different type of accident from any accident previously evaluated. Since the Technical Specification Bases Control Program represents an administrative function performed under existing regulatory controls, it too would not create the possibility of a new or different type of accident from any previously evaluated.

The addition of new programs which incorporate existing Technical Specification requirements and commitments will have no effect on the design or operation of the plant and would not create the possibility of a new or different type of accident from any previously evaluated.

A reporting function such as report submittals would not change the configuration or operation of the plant. Consequently, the elimination of the requirement to submit the Startup Report and the Special Report dealing with iodine activity levels, would not create the possibility of a new or different type of accident from any accident previously evaluated.

Since the operation or configuration of the plant is not changed by the type of personal dosimeter, this change would not create the possibility of a new or different type of accident from any accident previously evaluated.

Therefore, the proposed changes would not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

Relocating existing requirements to BGE-controlled documents, eliminating requirements which duplicate regulations, locating similar requirements within the same sections and making necessary editorial corrections to incorporate the proposed changes would not affect the Updated Final Safety Analysis Report design bases, accident analysis assumptions or any margin of safety described in the Technical Specification Bases. In addition, these proposed changes do not affect effluent release limits, monitoring equipment or practices. Therefore, these proposed changes would not involve a significant reduction in a margin of safety.

The use of an STA should provide an additional margin of safety in the accident response function of licensed operators beyond that considered in the accident analysis. Since the STA is required to have the same training and educational qualifications in either the individual or dual role, the use of a dual role STA should have minimal impact. Consequently, the proposed change would not involve a significant reduction in a margin of safety. The Technical Specification Bases Control Program is an administrative change controlling how Technical Specification basis information is reviewed and incorporated. Therefore, this change would not involve a significant reduction in a margin of safety.

The addition of new programs which incorporate existing Technical Specification requirements and commitments will have no effect on the design or operation of the plant and would not result in a significant reduction in the margin of safety.

Activities described in the Startup Report will continue to be performed and corrective action taken when required. Similarly, iodine activity levels will continue to be monitored and actions taken, including the issuance of a Licensee Event Report when conditions warrant. Considering the above, elimination of the two reporting requirements would have no impact on the margin of safety.

Plant operating parameters are not affected by the type of personnel monitoring device used and as a consequence, would not impact a margin of safety. Since the replacement dosimeter provides a more effective mechanism for estimating dose, there is no degradation in personal safety levels. Consequently, the proposed change would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

Attorney for licensee: Jay E. Silbert, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Ledyard B. Marsh

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment requests: September 17, 1993, as supplemented July 28, 1995

Description of amendment requests: As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station needed attention focused on the existing custom Technical Specifications (TS) used.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TS for both Dresden and Quad Cities against the Standard Technical Specifications (STS) contained in NUREG-0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing TS to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG-0123.

The TSUP for Dresden and Quad Cities is not a complete adoption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operation and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations.

The September 17, 1993, and July 28, 1995, applications proposed to upgrade only Section 3/4.5 (Emergency Core Cooling Systems) of the Dresden and Quad Cities TS.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.5 are based on STS guidelines or later operating BWR plants' NRC accepted changes. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and has been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems that make up the Emergency Core Cooling Systems are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. In addition, the proposed surveillance requirements for the proposed amendments to these systems are generally more prescriptive than the current requirements specified within the Technical Specifications. The additional surveillance requirements improve the reliability and availability of all affected systems and therefore, reduce the consequences of any accident previously evaluated as the probability of the systems outlined within Section 3/4.5 of the proposed Technical Specifications performing their intended function is increased by the additional surveillances.

Create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, the addition of requirements which are based on the current safety analysis, and some minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification Section 3/4.5 is based on STS guidelines or later operating BWR plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the STS or later operating BWRs. Any deviations from STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems that make up the Emergency Core Cooling Systems are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations. In addition, the proposed surveillance requirements for affected systems associated with the Emergency Core Cooling Systems are generally more prescriptive than the current requirements specified within the Technical Specifications; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Involve a significant reduction in the margin of safety because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, the addition of requirements which are based on the current safety analysis, and some minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the latter individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain with their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to Technical Specification Section 3/4.5 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden or Quad Cities Stations. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden or Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden or Quad Cities and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems associated with the Emergency Core Cooling Systems when required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: for Dresden, Morris Public Library, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690

NRC Project Director: Robert A. Capra

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: June 17, 1993, as supplemented July 5, 1995

Description of amendment request: The initial proposed amendment request dated June 17, 1993, was previously noticed in the Federal Register on July 21, 1993 (58 FR 39048). The proposed amendment would revise Technical Specification 5.3.1, ''Fuel Assemblies'' to provide flexibility in the repair of fuel assemblies containing damaged and leaking fuel rods by reconstituting the assemblies in accordance with the guidance in Generic Letter (GL) 90-02, Supplement 1, ''Alternative Requirements For Fuel Assemblies In The Design Features Section Of Technical Specifications,'' issued on July 31, 1992. The application is also generally consistent with the format and content of the improved Standard Technical Specifications for Westinghouse plants provided in NUREG-1431.

Additional information was submitted on July 5, 1995, that added TS changes to increase the fuel enrichment limit from 4.0 to 5.0 weight percent U-235 that were not previously included the initial June 17, 1993, amendment application. This additional information is being noticed to provide for public comment and opportunity for hearing.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration (58 FR 39048). The NRC staff's analysis of the July 5, 1995, supplement against the standards of 10 CFR 50.92(c) is presented below.

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

There is no increase in the probability or consequences of an accident in the new fuel vault since the only accident that would be affected by this change would be a criticality accident and it has been shown that the worst-case k_{eff} under optimum moderation conditions continues to be less than or equal to 0.98.

There is no increase in the probability of a fuel drop accident in the Spent Fuel Storage Pool since the mass of an assembly will not be significantly affected by the increase in fuel enrichment. The likelihood of other accidents, previously evaluated and described in Section 9.1.2 of the Final Safety Analysis Report (FSAR), is also not affected by the proposed changes. Since the increase in fuel enrichment will allow for extended fuel cycles, it could be postulated that there may be a decrease in fuel movement and the probability of an accident may likewise be decreased. There is also no increase in the consequences of a fuel drop accident in the Spent Fuel Pool since the fission product inventory of individual fuel assemblies will not change significantly as a result of increased initial enrichment. In addition, no change to safety-related systems is being made.

Therefore, the consequences of a fuel rupture accident remain unchanged. In addition, it has been shown that k_{eff} is less than or equal to 0.95, under all conditions. Therefore, the consequences of a criticality accident in the Spent Fuel Pool remain unchanged as well.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not create the possibility of a new or different kind of accident since fuel handling accidents (fuel drop and misplacement) are not new or different kinds of accidents. Fuel handling accidents are already discussed in the FSAR for fuel with enrichments up to 4.0 weight % and additional analyses have been performed for fuel with enrichment up to 5.00 weight %.

3.

The proposed changes do not involve a significant reduction in the margin of safety.

The proposed change does not involve a significant reduction in the margin of safety since, in all cases, a spent fuel pool k_{eff} less than or equal to 0.95 is being maintained. Criticality analyses have also been performed that show that the new fuel storage vault will remain subcritical under a variety of moderation conditions, from fully flooded to optimum moderation. As discussed above, the Spent Fuel Pool will remain sufficiently subcritical during any fuel misplacement accident.

Based on this analysis, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the supplemental amendment submittal involves no significant hazards consideration.

Local Public Document Room location:: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Duke Power Company, Docket Nos. 50-269, 50-270 and 50-287, Oconee Nuclear Station, Units 1, 2 and 3, Oconee County, South Carolina

Date of amendment request: July 26, 1995

Description of amendment request: The proposed amendments would provide a one-time extension of the allowable outage time from 72 hours to 7 days. This extension is necessary to implement a modification to the degraded grid protection system and the external grid trouble protection system.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:(1) Involve a significant increase in the probability or consequences of an accident previously evaluated:

Each accident analysis addressed within the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to the change proposed within this amendment request. The design basis of the auxiliary electrical systems is to supply the required engineered safeguards (ES) loads of one unit and the safe shutdown loads of the other two units. The systems are arranged so that no single failure will jeopardize plant safety.

The probability of any Design Basis Accident (DBA) is not significantly increased by this change. In addition, the consequences of the accidents are within the bounds of the FSAR analyses. The reliability of the emergency power system is not significantly affected by a one time extension of allowable outage time for the overhead power path. The underground power path is adequate to assure operability of the Oconee ES loads. Finally, the enhancement of the Degraded [Grid] Protection System will eliminate a concern which was expressed by the EDSFI audit team.

(2) Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:

Inoperability of the yellow bus is functionally equivalent to inoperability of the Keowee Main Step-up Transformer in that it renders the overhead emergency power path inoperable. The Keowee Main Step-up Transformer is allowed to be inoperable for a period not to exceed 28 days. This Technical Specification requirement for the

Keowee Main Step-up Transformer has been reviewed and approved by the NRC. Therefore, operation of ONS [Oconee Nuclear Station] in accordance with this Technical Specification amendment will not create any failure modes not bounded by previously evaluated accidents. Consequently, this change will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

(3) Involve a significant reduction in a margin of safety:

The design basis of auxiliary electrical systems is to supply the required ES loads of one Unit and safe shutdown loads of the other two units. The underground power path is adequate to ensure operability of the ES loads during the outage of the yellow bus. The reliability of the emergency power system is not significantly affected by a one time extension of allowable outage time for the overhead power path. Therefore, there will be no significant reduction in any margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Oconee County Library, 501 West South Broad Street, Walhalla, South Carolina 29691

Attorney for licensee: J. Michael McGarry, III, Winston and Strawn, 1200 17th Street, NW., Washington, DC 20036

NRC Project Director: Herbert N. Berkow

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412 Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: July 10, 1995

Description of amendment request: The proposed amendment would modify the technical specifications to minimize the potential for boron deletion of the reactor coolant system (RCS) during startup of an isolated loop. The changes would permit RCS loop isolation only during Modes 5 and 6. RCS loop isolation valves would be required open with power removed from each isolation valve operator during Modes 1, 2, 3, and 4. Primary grade water would be isolated from the RCS during Modes 4, 5, and 6, except during planned boron dilution or makeup activities.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed amendment would modify the method used to prevent an inadvertent boron dilution event during hot shutdown, cold shutdown and during refueling. An uncontrolled boron dilution transient cannot occur during this mode of operation. Inadvertent boron dilution is prevented by administrative controls which isolate the primary grade water system isolation valves from the Chemical and Volume Control System, except during planned boron dilution or makeup activities. Thus unborated water can not be injected into the reactor coolant system, making an unplanned boron dilution at these conditions highly improbable, since the source of unborated water to the charging pumps is isolated. This precludes the primary means for an inadvertent boron dilution event in this mode of operation.

The primary grade water system isolation valves may be opened when directed by the control room during this mode of operation only for a planned boron dilution or makeup activity. The primary grade water system isolation valves will be verified to be locked, sealed or otherwise secured in the closed position after the planned boron dilution or makeup activity is completed. During planned boron dilution events, operator attention will be focused on the boron dilution process and any inappropriate blender operation will be readily identified.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation supplied by the source range nuclear instrumentation. High count rate is alarmed in the reactor containment and the control room. In addition a high source range flux level is alarmed in the control room. The count rate increase is proportional to the subcritical multiplication factor.

The proposed amendment would also modify the method used to prevent an adverse reactor transient during startup of an isolated reactor coolant loop. Procedures require that the isolated loop water boron concentration be verified prior to opening loop isolation valves. Procedures also require an isolated loop to be drained and refilled from water supplied from the Refueling Water Storage Tank (RWST) or Reactor Coolant System (RCS) prior to opening either the hot or cold leg isolation valves. Using water from the RWST or RCS ensures 1) that the boron concentration of the isolated loop is sufficient to prevent a dilution of the active reactor coolant loops and reducing the shutdown margin to below those values used in safety analyses when the isolated loop is returned to service, and 2) that no single failure could cause an isolated loop to be filled with unborated water.

Thus procedures and interlocks prevent inadvertent opening of loop isolation valves and require that the startup of an isolated loop be performed in a controlled manner that virtually eliminates any sudden positive reactivity addition from boron dilution. Thus the core cannot be adversely affected by the startup of an isolated loop and fuel design limits are not exceeded. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not create the possibility of a new or different kind of accident. No new systems, structures or components are being proposed. Acceptable alternative administrative controls are being proposed to address inadvertent boron dilution and the startup of inactive reactor coolant loops.

The primary source of unborated water will be isolated from injecting by the charging pumps into the reactor coolant system during hot shutdown, cold shutdown, and refueling, except for planned boron dilution events and makeup activities. The proposed administrative controls prevent the possible accident previously evaluated, i.e., an inadvertent boron dilution event.

A currently installed interlock to recirculate reactor coolant in an isolated loop is proposed to be deleted. In its place, each reactor coolant isolated loop will be drained and refilled with water supplied from the RWST just before the loop is returned to service. This administrative control will prevent any inadvertent reactivity transient when returning the loop to service. Thus, the proposed administrative controls will prevent the type of accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed changes will continue to ensure that adequate protection is provided against an inadvertent boron dilution and the adverse effects from the startup of an isolated reactor coolant loop. General Design Criteria 10 requirements will not be exceeded with respect to demonstrating specified acceptable fuel design limits. The required indications and functions are still maintained in accordance with current technical specification requirements and the shutdown margin is unaffected. Therefore, the proposed change will not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: B. Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

Duquesne Light Company, et al., Docket No. 50-334, Beaver Valley Power Station, Unit 1, Shippingport, Pennsylvania

Date of amendment request: July 11, 1995

Description of amendment request: The proposed amendment would revise the required area of the Reactor Coolant System (RCS) overpressure protection system vent from 3.14 square inches to 2.07 square inches. This vent is provided to relieve a potential RCS overpressure condition if the power-operated relief valves (PORVs) are not operable. The proposed vent area is equal to the relief area of a PORV. A single PORV is capable of providing sufficient relief capacity to mitigate potential low temperature overpressurization events.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change is considered to be editorial since it replaces the 3.14 square inch vent size stated in overpressure protection system (OPPS) Specifications 3.4.9.3, 3.1.2.1.b, and 3.1.2.3 and Bases 3/4.1.2 and 3/4.4.9 with a 2.07 square inch vent size. This ensures the vent size stated in the technical specifications is consistent with the actual size of an installed PORV. These changes maintain consistency with the analyses assumptions and the operation of the OPPS in accordance with applicable analyses and the UFSAR [Updated Final Safety Analyses Report]. Therefore, we have concluded that these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve any physical changes to the OPPS or their setpoints. These changes do not change any function previously provided by the OPPS. These changes do not affect any failure modes defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of these changes. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR.

3. Does the change involve a significant reduction in a margin of safety?

The proposed changes will not affect the operation of or the reliability of the OPPS. These changes do not affect the manner in which the plant is operated or involve a change to equipment or features that affect the operational characteristics of the plant. Therefore, operation of the plant in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of amendment request: July 20, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.8.1.1 to incorporate guidance provided in NRC Generic Letter (GL) 84-15, ''Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability,'' and GL 93-05, ''Line-Item Technical Specification Improvements To Reduce Surveillance Requirements For Testing During Power Operation,'' which includes (1) revised requirements for testing the operable emergency diesel generators (EDGs) for various combinations of inoperable offsite circuits and EDGs and (2) revised surveillance requirements for the EDGs. The revised surveillance requirements include specifying generator voltage, frequency limits, and diesel starting time. In addition, several editorial changes would be made to TS 3/4.8.1.1 which would be consistent with the guidance provided in the NRC's Improved Standard Technical Specifications (NUREG-1431).

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The probability of occurrence of a previously evaluated accident is not increased because the allowable outage times for the offsite circuits and diesel generators remain unchanged. The consequences of an accident previously evaluated is not increased because reducing the diesel generator test frequency and permitting additional test evolutions are intended to minimize diesel wear and mechanical stress. By eliminating excessive testing, which can lead to premature diesel failures and minimizing diesel wear and mechanical stress, the diesel generator reliability is increased. The consequences of an accident previously evaluated is also not increased because the addition of the parameters for generator voltage, frequency, and diesel starting time to the surveillance requirement will provide additional assurance that the diesel generators are performing as assumed in the safety analysis. This proposed change does not affect the availability or reliability of the offsite circuits.

Therefore, this change will not increase the probability or consequences of an accident previously evaluated due to the continued availability and reliability of the A.C. electrical power sources.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not alter the method of operating the plant. The changes do not introduce any new failure modes and are intended to increase the diesel generator reliability and provide additional assurance that the diesels are performing as assumed in the safety analysis. The revision to the various action statements and surveillance requirements provide assurance that the diesel generators will be able to power their respective safety systems if required. The proposed changes do not impact the performance of any safety system.

Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The margin of safety is not reduced because the A.C. electrical power sources will continue to provide sufficient capacity, capability, redundancy, and reliability to ensure availability of necessary power to engineered safety feature (ESF) systems. The ESF systems will continue to function, as assumed in the safety analyses, to ensure that the fuel, reactor coolant system and containment design limits are not exceeded. The elimination of excessive testing on the diesel generators are permitting additional test evolutions, which result in less diesel wear and mechanical stress, are intended to increase diesel reliability. The increased reliability of the diesels adds to the ability of the A.C. electrical power source to provide power to ESF systems. The proposed additions to the surveillance requirements will provide additional assurance of the ability of the A.C. electrical power sources to provide power to ESF systems.

Therefore, this proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

Duquesne Light Company, et al., Docket No. 50-412, Beaver Valley PowerStation, Unit 2, Shippingport, Pennsylvania

Date of amendment request: July 24, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.4.11, ''Relief Valves,'' and associated Bases to make Unit 2 TS 3/4.4.11 consistent with Unit 1 TS 3/4.4.11, which was revised by Unit 1 License Amendment No. 187 issued on May 15, 1995. The proposed amendment would also generally reflect the guidance provided in NRC Generic Letter 90-06 and in the NRC's Improved Standard Technical Specifications (NUREG-1431).

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Implementation of these changes will increase the availability of the power-operated relief valves (PORVs) and their associated block valves. The increased availability is obtained through maintaining power to the block valves which are closed to control PORV seat leakage. Maintaining power to the block valve provides the flexibility of reopening the valves to control reactor coolant system pressure. The proposed change modifies Specification 3.4.11 actions, a surveillance requirement, and Bases to generally reflect the requirements of Generic Letter (GL) 90-06, and the guidance provided in NUREG-1431, ''Improved Standard Technical Specifications'' (ISTS) and is consistent with the changes the NRC approved for Unit No. 1. A revised stress analysis has been completed that takes credit for the speed at which the block valve opens when manually reducing reactor coolant system pressure. The block valve relatively slow opening speed reduces the peak pressure surge and results in acceptable downstream piping stress values. The PORV downstream piping has been evaluated assuming manual vent path operation with cold loop seal slug flow and it has been determined that the piping supports can accept these design transient loads. The proposed change to the action statement to close the block valve to isolate a PORV and maintain power to the block valve does not significantly increase the probability of a small break loss of coolant accident. No PORV function has been deleted and the PORV and block valve continue to be capable of being manually closed at any time. As a result of the change to action ''a,'' an exception to the stroking requirements is no longer required, therefore, reference to action ''a'' in Surveillance Requirement 4.4.11.2 has been deleted. Closing the block valve for a PORV that is not capable of being manually cycled and removing power to the block valve assures that the valve will not be inadvertently opened when the condition of the PORV is uncertain.

The changes remain consistent with the analysis assumptions regarding the operation of the PORVs and block valves and provides increased assurance of their availability in mitigating the consequences of a steam generator tube rupture (SGTR) accident. The requirements of GL 90-06 are substantially addressed in the ISTS which have been incorporated here except for specific design differences. Minor editorial changes involving capitalization have been incorporated to maintain the format and content and do not affect any of the requirements, the accident analyses, or the operation of the plant. Therefore, we have concluded that these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR [Updated Final Safety Analysis Report].

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes to the action statements for the PORVs and the associated block valves will improve the availability of these valves for normal operation and for mitigation of a SGTR accident. The proposed changes do not involve any physical changes to the PORVs or their setpoints. These changes do not delete any design basis accident function previously provided by the PORV vent path nor has the probability of inadvertent opening been increased. Accordingly, no new limiting single failure has been identified as a result of these changes. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR.

3. Does the change involve a significant reduction in a margin of safety?

The proposed changes have been incorporated to provide the capability to manually stroke the vent path using the block valve to control the pressure surge as a PORV opens. The resultant downstream piping forces were found acceptable, therefore, power can be maintained to the block valve when the block valve has been closed to isolate a PORV because of excessive seat leakage. This will allow operation of the PORVs in a manner similar to the guidance provided in GL 90-06 to improve PORV availability. These changes will improve the operator use of an isolated PORV since it is now analyzed to be manually cycled with the block valve closed and power maintained so the operator can use the PORV if required to mitigate the effects of a SGTR accident. This is consistent with the intent of the ISTS and does not affect the UFSAR, therefore, operation of the plant in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 1500l.

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

Entergy Operations, Inc., Docket No. 50-368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: April 4, 1995

Description of amendment request: The proposed amendment revises the minimum water level that is required to be maintained over irradiated fuel assemblies during latching and unlatching of control element assemblies.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The fuel handling accident analysis assumes that a fuel assembly is dropped during fuel handling. During the latching and unlatching of the CEAs, the upper guide structure is in place and the CEDM extension shaft assemblies are disconnected from their CEA for subsequent removal with the vessel upper guide structure. The dropping of a CEA from the maximum height of six inches will not damage that particular fuel assembly or any surrounding fuel assemblies since this movement is confined to within the upper guide structure and the guide tubes of the associated fuel assembly during this activity. This less than six inches of movement does not have the potential to result in a fuel handling accident; therefore, an increase in the probability of this accident does not occur. The requirement to have at least 23 feet of water over the top of the irradiated fuel assemblies during fuel and CEA movement ensures that, should a fuel handling accident occur, the resulting offsite dose consequences are mitigated. The six inch movement of the CEA during CEA decoupling does not constitute fuel or CEA movement which would result in a fuel handling accident. As such, Technical Specifications are unchanged with respect to the mitigating requirements for a fuel handling accident.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change does not change the design, configuration, or method of operation of the plant; therefore, it does not create the possibility of a new or different kind of accident. Because no new equipment is being introduced, and no equipment is being operated in a manner inconsistent with its design, the possibility of equipment malfunction is not increased. The proposed change adds an exception to the applicability section and is bounded by the existing fuel handling accident analysis.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

There is no reduction in margin of safety in that 23 feet of water is still maintained over the irradiated fuel assemblies anytime there is a potential for a fuel handling accident. Adding the exception of the latching and unlatching of the CEAs to the applicability section does not involve a change in the accident analysis for fuel handling which remains bounding.

Therefore, this change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

Entergy Operations, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: July 21, 1995

Description of amendment request: The proposed change requests that the current expiration date for license NPF-29 be changed to reflect the issuance date of the new license granted Grand Gulf on November 1, 1984. The change consists of extending the expiration date to 40 years from the date of issuance of license NPF-29 (November 1, 1984 to November 1, 2024).

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

a. No significant increase in the probability or consequences of an accident previously evaluated results from this change.

The proposed change does not affect the design or operation of any plant system. The effect of 40 years of full power operations has previously been evaluated and documented in the Updated Final Safety Analysis Report (UFSAR). The design life of structures, systems and components is controlled by existing plant problems [sic., programs] and processes that are not affected by this change. The proposed change will simply allow Grand Gulf to achieve its original planned 40 years of service. Equipment associated with initiating event frequencies or accident mitigation must continue to meet all applicable maintenance and operability requirements regardless of license duration (It is also interesting to note that the license duration limitation of 40 years, as contained in 10 CFR 50.51 is not a limitation resulting from concerns over plant aging effects. ''In fact, the limit was a compromise between the efforts of the Justice Department and electric cooperatives, who championed a 20-year limit on the basis of antitrust concerns, and the view of the utility industries that a longer period was necessary to ensure full amortization of a nuclear power plant.'' (56 FR 64961, December 13, 1991)). Therefore, the probability or consequences of previously analyzed accidents are not significantly increased.

b. The change would not create the possibility of a new or different kind of accident from any previously analyzed.

The proposed change will not add any plant equipment or introduce any new modes of plant operation. The change will only amend the operating license to allow 40 years of full power operations. The proposed change does not affect the current maintenance or surveillance practices, which are designed to maintain and monitor the current service life of plant structures, systems and components in accordance with regulatory requirements. Therefore, the proposed change does not create the possibility of new equipment failure modes or a new or different kind of accident from any accident previously evaluated.

c. The change would not involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety since it only provides for 40 years of full power operations for which the plant is designed. Current Technical Specification surveillance requirements (e.g. associated with 10 CFR 50 Appendix H) and other regulatory requirements remain in place and will ensure continued compliance with applicable safety margins.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, Mississippi 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005-3502

NRC Project Director: William D. Beckner

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: June 20, 1995

Description of amendment request: The proposed Technical Specifications (TS) changes would remove the surveillance interval text for the 10 CFR Part 50, Appendix J, Type A test (Integrated Leak Rate Test or ILRT), and Drywell-to-Suppression Chamber (bypass) leakage test specified in TS Surveillance Requirements (SR) 4.6.1.2.a, 4.6.1.2.b, and 4.6.2.1.e.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed TS changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The primary containment and the suppression chamber are not considered to be accident initiators, they are accident mitigators. There are no physical or operational changes to the containment or suppression structure, system or components being made as a result of the proposed changes. These changes will not impose different requirements and adequate control of information will be maintained. These TS changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, the proposed TS changes to eliminate the details of the test intervals will not increase the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes remove the specific surveillance test interval text from TS and address the interval by direct reference to the applicable regulation. The proposed TS changes do not make any physical or operational changes to existing plant systems or components. Furthermore, the primary containment and suppression chamber act as accident mitigators not initiators. Therefore, the possibility of a new or different kind of accident than from any accident previously evaluated is not introduced.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

LGS [Limerick Generating Station] TS Bases 3/4 6.1.2 state that surveillance testing is consistent with 10 CFR 50, Appendix J and does not specify a SR test interval. TS Bases 3/4 6.2, describing the bypass test does not specify a SR test interval. However, the NRC Safety Evaluation related to amendment Nos. 68 (Unit 1) and 31 (Unit 2) concluded that it is acceptable for the drywell-to-suppression chamber test frequency to coincide with the 10 CFR 50, Appendix J, Type A test, since individual vacuum breaker leakage tests are an acceptable alternative to an integrated suppression pool bypass test during outages for which a Type A containment integrated leak rate test is not conducted. The alternative bypass test requirement, TS SR 4.6.2.1.f, is not affected by these changes.

The Type A test, and bypass SR test intervals are adequately presented in the test implementing procedures, and TS will directly reference 10 CFR 50, Appendix J, for the appropriate test interval.

Therefore, the proposed TS changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

Power Authority of The State of New York, Docket No. 50-286, Indian PointNuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: July 21, 1995

Description of amendment request: The proposed amendment would change Technical Specifications Section 6.0 (Administrative Controls) to replace the title-specific list of members on the Plant Operating Review Committee (PORC) with a more general statement of membership requirements. The scope of disciplines represented on the PORC would also be expanded to include nuclear licensing and quality assurance. The proposed amendment would also change the title ''Resident Manager'' to ''Site Executive Officer.'' This title change would not affect the reporting relationship, authority, or responsibility of the position.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Operation of the Indian Point 3 Nuclear Power Plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative in nature and do not involve plant equipment or operating parameters. There is no change to any accident analysis assumptions or other conditions which could affect previously evaluated accidents. The proposed changes will not decrease the organization's ability to respond to a design basis accident.

2. Create the possibility of a new or different kind of accident from those previously evaluated.

Since the proposed changes are administrative in nature and do not involve hardware design, modifications or operation, the possibility of new or different accidents is not created.

3. Involve a significant reduction in the margin of safety.

The proposed title change for the Resident Manager is an administrative change and does not affect the responsibilities, authority, or reporting relationships for this management position. Replacing the title specific list of PORC members with a statement of membership requirements for the committee does not reduce the effectiveness of the committee to advise the Resident Manager (Site Executive Officer) on matters regarding nuclear safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: Ledyard B. Marsh

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: March 30, 1995

Description of amendment request: The proposed change to the Technical Specifications (TS) would change TS Table 3.3.1-2, ''Reactor Protection System Response Times'', TS Table 3.3.2-3, ''Isolation System Instrumentation Response Time'', TS Table 3.3.3-3, ''Emergency Core Cooling System Response Times'', and associated Bases. The proposed changes to the above-referenced TS Tables would eliminate the requirement to perform response time testing for certain classes of equipment.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The purpose of the proposed Technical Specification change is to eliminate response time testing requirements for selected instrumentation in the Reactor Protection System, Isolation System, and Emergency Core Cooling System. However, because of the continued application of other existing Technical Specification requirements such as channel calibrations, channel checks, channel functional tests, and logic system functional tests, the response time of these systems will be maintained within the acceptance limits assumed in plant safety analyses and required for successful mitigation of an initiating event. The proposed Technical Specification changes do not affect the capability of the associated systems to perform their intended function within their required response time.

The BWR Owners' Group has completed an evaluation (NEDO-32291, ''System Analyses for the Elimination of Selected Response Time Testing Requirements'') which demonstrates that response time testing is redundant to the other Technical Specification requirements listed in the preceding paragraph. These other tests are sufficient to identify failure modes or degradation in instruments response time and ensure operation of the associated systems within acceptance limits. There are no known failure modes that can be detected by response time testing that cannot be detected by the other Technical Specification tests. Hope Creek Generating Station is specifically bounded by the assumptions and justifications in General Electric Company Licensing Topical Report, NEDO-32291, ''System Analyses for Elimination of Selected Response Time Testing Requirements.''

2. Will not create the possibility of a new or different kind of accident from any accident previously evaluated.

As discussed above, the proposed Technical Specification changes do not affect the capability of the associated systems to perform their intended function within the acceptance limits assumed in plant safety analyses and required for successful mitigation of an initiating event. The proposed elimination of response time testing would not result in any new equipment, operating modes, or plant configurations.

3. Will not involve a significant reduction in a margin of safety.

The current Technical Specification response times are based on the maximum allowable values assumed in the plant safety analyses. These analyses conservatively establish the margin of safety. As described above, the proposed Technical Specification changes do not affect the capability of the associated systems to perform their intended functions within the allowed response time used as the basis for the plant safety analyses. Plant and system response to an initiating event will remain in compliance within the assumptions of the safety analyses, and therefore the margin of safety is not affected.

Although not explicitly evaluated, the proposed Technical Specification changes will provide an improvement to plant safety and operation by:

a) Reducing the time safety systems are unavailable

b) Reducing safety system actuations

c) Reducing shutdown risk

d) Limiting radiation exposure to plant personnel

e) Eliminating the diversion of key personnel to conduct unnecessary testing.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

Attorney for licensee: M. J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005-3502

NRC Project Director: John F. Stolz

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: April 18, 1995

Description of amendment request: The proposed changes to the Technical Specifications (TS) would change TS Table 4.3.7.1-1 ''Radiation Monitoring Instrumentation Surveillance Requirements.'' This change would increase the channel functional test interval from monthly to quarterly for each instrument.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change involves no hardware changes, no changes to the operation of any systems or components, and no changes to existing structures. Increasing the interval between channel functional tests for the radiation monitoring instrumentation represent changes that do not affect plant safety and do not alter existing accident analyses.

2. Will not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change is procedural in nature concerning the channel functional test frequency for the radiation monitoring instrumentation not already on a quarterly surveillance. The channel functional test methodology for these instruments remains unchanged. The proposed changes, while slightly increasing the possibility of an undetected instrument error, will not create a new or unevaluated accident or operating condition.

3. Will not involve a significant reduction in a margin of safety.

The proposed change is in accordance with recommendations provided by the NRC regarding the improvement of Technical Specifications. These changes will result in perpetuation of current safety margins while reducing regulatory burden and decreasing equipment degradation.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

Attorney for licensee: M. J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005-3502

NRC Project Director: John F. Stolz

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of amendment request: May 4, 1995

Description of amendment request: The proposed change to the Technical Specifications (TS) would change TS 3/4.6.1.8, ''Drywell and Suppression Chamber Purge System'', to increase the annual operational limit for the drywell and suppression chamber purge system from 120 to 500 hours.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change involves no hardware changes and no changes to existing structures. Increasing the annual operational limit of the drywell and suppression chamber purge system will not increase the probability of a loss-of-coolant accident. While increased usage of the purge system will result in a slight increase in the possibility that these valves will be open during a LOCA, it will not alter or impact previous LOCA analyses.

2. Will not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change will not result in an unanalyzed condition. While the increase in purge system operation will slightly increase the possibility of the containment vent and purge valves being open at the onset of a LOCA event, the valves have been established as capable of isolating the containment within five seconds. This is well within the bounds of existing LOCA analyses which assume an open duration of 175 seconds. Therefore, this change will not require a new or different accident analysis.

3. Will not involve a significant reduction in a margin of safety.

The proposed change will not alter existing systems, equipment, components, or structures. The method of operating the drywell and suppression chamber purge system will not be altered by the increased annual usage. While there is a slight increase in the possibility of purge operations at the onset of a LOCA, any resulting release would be insignificant and bounded by existing LOCA analyses. Operation of the drywell and suppression chamber purge system based on these proposed changes will remain within the guidance provided in the NRC's Branch Technical Position CSB 6-4.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

Attorney for licensee: M. J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW., Washington, DC 20005-3502

NRC Project Director: John F. Stolz

Saxton Nuclear Experimental Corporation (SNEC), Docket No. 50-146, Saxton Nuclear Experimental Facility (SNEF), Bedford County, Pennsylvania

Date of amendment request: June 2, 1995, as supplemented on June 23, 1995.

Description of amendment request: The proposed changes to the technical specifications are administrative in nature. The proposed amendment would revise the organization structure associated with the SNEF to allow General Public Utilities Nuclear Corporation resources to be applied to SNEC activities within their normal organizational structure; eliminating the need to identify and compartmentalize a portion of the organization as specific to SNEC. The proposed amendment would also revise the description and drawing of the SNEF site to reflect multiple gates in the SNEF fence.

Basis for proposed no significant hazards considerationDetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below: The proposed changes do not involve a significant hazards considerations because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The administrative changes will not impact the physical condition of the containment vessel as it relates to the risk of fire, flood or radiological hazard.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

In its present condition, the only accidents applicable to the site are those addressed above.

3. Involve a significant reduction in a margin of safety.

The proposed administrative changes would have no effect on any margins of safety for any evaluated accidents.

The NRC staff has reviewed the analysis of the licensee and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Saxton Community Library, 911 Church Street, Saxton, Pennsylvania 16678Attorney for the Licensee: Ernest L. Blake, Jr., Esquire, Shaw, Pittman, Potts, and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: Seymour H. Weiss

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: June 30, 1995

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) for the pressurizer power operated relief valves (PORVs) to follow the guidance of Generic Letter (GL) 90-06, Generic Issue 70, and the improved Westinghouse Standardized Technical Specifications (NUREG-1431, Rev. 1).

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The probability or consequences of an accident previously evaluated is not significantly increased.

There is no increase in the probability of an accident because the physical characteristics of the PORVs and their block valves remain unchanged. No changes to any hardware or software that affects these components is planned.

The PORVs are pressure relieving devices and only two failure modes need to be considered. The first is that one or more PORVs or block valves fail to open when required. This is not

a significant concern and is not a credible cause of any accident. The second mode is failing to close which includes depressurization of the RCS [reactor coolant system] and a reactor trip on low pressurizer pressure or overtemperature [delta]T. The consequences for the more limiting Pressurizer Safety Valve Accidental Depressurization event has been analyzed with acceptable results.

There is no increase in the consequences of an accident as a result of this change, because only one PORV is required to mitigate the consequences of a design basis Steam Generator Tube Rupture. There is sufficient redundancy to ensure one PORV is available to perform this function even if one PORV is inoperable or incapable of being manually cycled. The validation of the Emergency Operating Procedures on the VCSNS [Virgil C. Summer Nuclear Station] simulator demonstrated that one pressurizer PORV has sufficient capacity to depressurize the RCS in a time frame which will not cause the offsite doses presented in the FSAR [Final Safety Analysis Report] to be exceeded.

The PORVs are utilized to depressurize the RCS and equalize the pressure between the primary and secondary systems. This stops the intrusion of RCS water into the secondary which can be released into the atmosphere. By the time the PORVs are called upon, the affected steam generator (SG) has been identified and steps have been taken to isolate the faulted SG. This acts to minimize the radiological impact on the health and safety of the public. In all cases, the dose results are within 10 CFR 100 limits.

2. The possibility of an accident or a malfunction of a different type than any previously evaluated is not created.

The proposed TSCR [TS Change Request] does not involve any physical changes to the plant or decrease the number of PORVs and block valves that must be capable of performing their intended function. These components are used to mitigate the effects of postulated events and their failure has already been considered. The worst case failure, either not opening or not closing, has been evaluated and is bounded by other more limiting accidents.

3. The margin of safety has not been significantly reduced.

The currently approved TS permits all three PORVs and/or their block valves to be inoperable as long as precautions are taken to assure that RCS would not leak-by, assuming single failures and spurious operation. The proposed TSCR would require a minimum of two PORVs and block valves to be operable, or at least capable of being manually cycled, in Modes 1, 2, and 3. This is in fact an increase in margin and provides for greater reliability with the added benefit that the probability of challenges to the pressurizer code safety valves will be lessened.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Project Director: Frederick J. Hebdon

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: July 28, 1995

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to exclude the requirement to perform the slave relay test of the 36-inch containment purge supply and exhaust valves on a quarterly basis while the plant is in Modes 1, 2, 3, or 4.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

No, the probability or consequences of an accident previously evaluated would not be increased since no credit is taken for the valves in FSAR [Final Safety Analysis Report] Chapter 15.

The only credible accident discussed in FSAR Chapter 15 that applies to these valves is a fuel handling accident inside containment (15.4.5.1). The analysis assumes the escaped gases are released instantaneously to the environment via the Reactor

Building purge system. The analysis does not take credit for these valves nor for filtration or holdup time during release. The result of the analysis is acceptable and offsite doses are within the limits of 10 CFR 100.

TS 3.6.1.7 requires that these valves be sealed shut during Modes 1, 2, 3, and 4. When sealed shut, these valves will not open via any signal.

With these valves already in a shut position, neither the probability nor the consequences of an accident are increased.

2. Does the change create the possibility of a new or different kind of accident from any previously evaluated?

No, the 36'' [inch] containment purge exhaust and supply valves will not be placed in a condition different from that evaluated previously.

The only credible accident discussed in FSAR Chapter 15 that applies to these valves is a fuel handling accident inside containment (15.4.5.1). The analysis assumes the escaped gases are released instantaneously to the environment via the Reactor Building purge system. The analysis does not take credit for these valves nor for filtration or holdup time during release. The result of the analysis is acceptable and offsite doses are within the limits of 10 CFR 100.

Additionally, TS 3.6.1.7. requires that these valves be sealed shut during Modes 1, 2, 3, and 4. When sealed shut, these valves will not open via any signal.

3. Does the change involve a significant reduction in the margin of safety?

TS 4.3.2.1. requires that this slave relay test be performed quarterly. This surveillance is accomplished for the 36'' [inch] containment purge exhaust and supply valves by cycling the respective K615 relay. This will not provide assurance that the valve will perform its safety function since the valve is sealed closed. The proposed change will exclude the requirement to perform the K615 relay test (auto actuation logic and actuation relays - slave relay test) on a quarterly basis while the plant is in Modes 1, 2, 3,or 4.

TS 3.6.1.7. requires that these valves be sealed shut during Modes 1, 2, 3, and 4. When sealed shut, these valves will not open via any signal. Since this relay would not be needed to supply a signal to place these valves in the closed position, the margin of safety is not affected.

Based on the preceding analysis, SCE&G has determined that this change does no involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Project Director: Frederick J. Hebdon

Tennessee Valley Authority, Docket Nos. 50-259, 50-260 and 50-296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: June 2, 1995 (TS 353)

Description of amendment request: The proposed amendment supports replacement of the existing power range neutron monitoring equipment and implements ARTS/MELLL [average power range monitor and rod block monitor technical specifications/ maximum extended load line limit] analysis improvements.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Group A Changes: This proposed TS change is associated with the NUMAC PRNM [nuclear measurement analysis and control power range neutron monitor] retrofit design. The proposed TS change involves modification of the LCOs [limiting condition for operations] and SRs [surveillance requirements] for equipment designed to mitigate events which result in power increase transients. For the APRM [average power range monitor] system mitigative action is to block control rod withdrawal or initiate a reactor scram which terminates the power increase when setpoints are exceeded. For the RBM [rod-block monitor] system mitigative action is to block continuous control rod withdrawal prior to exceeding the MCPR [minimum critical power ratio] safety limit during a postulated Rod Withdrawal Error [RWE]. The worst case failure of either the APRM or the RBM systems is failure to initiate mitigative action (failure to scram or block rod withdrawal). Failure to initiate mitigative action will not increase the probability of an accident. Thus, the proposed change does not increase the probability of an accident previously evaluated.

For the APRM and the RBM systems, the NUMAC PRNM design, together with revised operability requirements (LCOs) and revised testing requirements (SRs), results in equipment which continues to perform the same mitigation functions under identical conditions with reliability equal to or greater than the equipment which it replaces. Because there is no change in mitigation functions and because reliability of the functions is maintained, the proposed change does not involve an increase in the consequences of an accident previously evaluated.

Group B Changes: This proposed change is associated with implementation of the ARTS/ MELLL analysis. The proposed change will permit expansion of the current allowable power/flow operating region and will apply a new methodology for assuring that fuel thermal and mechanical design limits are satisfied. Reference 3 evaluates operation in the MELLL region with assumed implementation of the ARTS changes. The conclusion of reference 3 is that for all events and parameters considered there is adequate design margin for operation in the MELLL region. Because operation in the MELLL region maintains adequate design margin, the proposed change does not significantly increase the probability of an accident previously evaluated.

In support of operation in the MELLL region, the proposed change modifies flow-biased APRM scram and rod block setpoints and implements new RBM power-biased setpoints. This potentially changes the way in which the APRM and RBM systems perform their mitigation functions. However, no credit for the flow-biased APRM scram or rod block is taken in mitigation of any design basis event; thus, changing the APRM setpoints does not impact the consequences of any accident previously evaluated. The proposed changes to the RBM system potentially impact mitigation of the RWE. However, per discussion in reference 3, the proposed RBM changes will assure that the RWE is not a limiting event; thus, the consequences of the RWE are not increased. The proposed change does not increase the consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes (Group A and Group B) involve modification and replacement of the existing power range neutron monitoring equipment, modification of the setpoints and operational requirements for the APRM and RBM systems, implementation of a new methodology for administering compliance with fuel thermal limits, and operation in an extended power/flow domain. These proposed changes do not modify the basic functional requirements of the affected equipment, create any new system interfaces or interactions, nor create any new system failure modes or sequence of events that could lead to an accident. The worst case failure of the affected equipment is failure to perform a mitigation action, and failure of this mitigative equipment does not create the possibility of a new or different kind of accident. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

Group A Changes: This proposed TS change is associated with the NUMAC PRNM retrofit design. The NUMAC PRNM change does not impact reactor operating parameters nor the functional requirements of the power range neutron monitoring system. The replacement equipment continues to provide information, enforce control rod blocks and initiate reactor scrams under appropriate specified conditions. The proposed change does not revise any safety margin requirements. The replacement APRM/RBM equipment has improved channel trip accuracy compared to the current system and meets or exceeds system requirements previously assumed in setpoint analysis. Thus, the ability of the new equipment to enforce compliance with margins of safety equals or exceeds the ability of the equipment which it replaces. The proposed change does not involve a reduction in a margin of safety.

Group B Changes: This proposed change is associated with implementation of recommendations presented in the ARTS/MELLL analysis. Operation in the MELLL region does not affect the ability of the plant safety-related trips or equipment to perform their functions, nor does it cause any significant increase in offsite radiation doses resulting from any analyzed event. Analyses documented in reference 3 demonstrate that for operation in the MELLL region adequate margin to design limits is maintained. Implementation of the ARTS improvements provides flow- and power-dependent thermal limits which maintain existing margins of safety in normal operation, anticipated operational occurrences and accident events. Implementation of power-biased RBM setpoints improves the margin of safety in a postulated RWE by assuring that the RWE is not a limiting event. The proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Athens Public Library, South Street, Athens, Alabama 35611

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET llH, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

Tennessee Valley Authority, Docket Nos. 50-259, 50-260 and 50-296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: June 8, 1995 (TS 361)

Description of amendment request: The proposed amendment clarifies the definition of operability for the RHRSW system standby coolant supply capability and revises the instrument numbers for several instruments that have been upgraded.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to TS 3.5.C.3 clarifies the operability requirements of the standby coolant supply capability. It does not change or degrade the nuclear safety characteristics of the RHRSW and RHR systems and will not affect the intent of the TS. The operation of the standby coolant supply capability is not a precursor to any design basis accident or transient analyzed in the BFN FSAR. The proposed changes to instrument numbers are administrative changes for the upgraded drywell temperature and pressure instrumentation. The proposed changes do not affect the design basis or the safety functions of the Primary Containment system, since the function and instrumentation range is not changed. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created by this change. The change to TS 3.5.C.3 adds the indication of associated valves of the function involved and a clarification of operability for the standby coolant supply connection to be commensurate with the RHR cross-connect capability. The proposed changes to instrument numbers are administrative changes effected by the upgrade of instrumentation. There are no automatic actions affected or compromised by these changes.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change to TS 3.5.C.3 does not affect any acceptable limit of operation or analysis assumption in the TS or Bases. The changes affect neither setpoints, calibration intervals, nor functional test intervals. The change does not affect any acceptable limit of operation or analysis assumption found in the TS or their bases. The proposed administrative changes to the instrument numbers do not affect the setpoint, calibration interval or function of the instrumentation. These changes do not affect any limiting conditions of operation or analysis assumption in the TSs or their bases. Therefore, the change does not reduce the margin of safety as defined in the basis for any TS.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Athens Public Library, South Street, Athens, Alabama 35611

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET llH, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

Tennessee Valley Authority, Docket Nos. 50-259, 50-260 and 50-296, Browns Ferry Nuclear Plant, Units 1, 2 and 3, Limestone County, Alabama

Date of amendment request: June 16, 1995 (TS 360)

Description of amendment request: The proposed change will revise the BFN Units 1, 2, and 3 Technical Specifications (TS) to permit the Traversing In-Core Probe (TIP) system to be considered operable with less than five TIP machines operable. The proposed amendment will allow the utilization of substitute data in lieu of data from inaccessible TIP measurement locations. The substitute data will be derived from either symmetric TIP measurement locations (under certain core conditions) or from normalized TIP data as calculated by the on-line core monitoring system.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The TIP system is not used to prevent, or mitigate the consequences of any previously analyzed accident or transient; nor are any assumptions made in any accident analysis relative to the operation of the TIP system. The primary containment isolation function (TIP withdrawal) is not affected. The

proposed TS change does not alter the fundamental process involved in calibrating neutron instrumentation (LPRMs) [local power range monitors], but requires that only the equipment associated with the TIP channels necessary for recalibrating LPRMs and for core monitoring functions be operable. Collection and storage of TIP data without using all TIP channels is acceptable because TIP machine normalization factors are ultimately derived from the most recent full core TIP set, which intercalibrates the TIP machines in a common core location.

Additionally, the use of symmetric detectors and analytical values as substitute data for inaccessible TIP channels does not compromise the ability of the process computer to accurately represent the spatial neutron flux distribution of the reactor core.

The core monitoring methodology is presently based on symmetry of rod patterns and fuel loading. This is not changed but extended to use a higher order of symmetry (octant symmetry) which exists with ''type A'' sequence rod patterns. Therefore, this change does not increase the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not involve the installation of any new equipment, or the modification of any equipment designed to prevent or mitigate the consequences of accidents or transients. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The total core TIP reading uncertainties will remain within the assumptions of the licensing basis. Therefore, the margin of safety to the MCPR [minimum critical power ratio] safety limits is not reduced. The ability of the process computer to accurately represent the spatial neutron flux distribution for the reactor core is not compromised. Additionally, the computer's ability to accurately predict the LHGR [linear heat generation rate], APLHGR [average planar linear heat generation rate], MCPR and its ability to provide for LPRM calibration is not compromised. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Athens Public Library, South Street, Athens, Alabama 35611

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET llH, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: October 21, 1994

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 3/4.6.1.2, ''Primary Containment Leakage.'' The changes would clarify that the main steam line isolation valves leakage is accounted for separately from the integrated primary containment leak rate or combined local leak rate results. Also, two references would be deleted, the test duration for use of Bechtel Corporation Topical Report BN-TOP-1 would be clarified, and the requirement to perform the third integrated leak rate in each 10-year service period in conjunction with the 10-year plant inservice inspection would be deleted. Exemptions to 10 CFR Part 50 Appendix J, ''Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,'' are also being requested in conjunction with the proposed TS changes.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

Part A - Formalize the Approval for Excluding the Main Steam Line Isolation Valve Leakages from Inclusion in i) the Overall Integrated Primary Containment Leak Rate and ii) the Combined Local Leak Rate, and Clarify that the Main Steam Lines are Not Required to be Vented and Drained for Type A Testing

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Since Appendix J was originally envisioned, alternative means of meeting the intent of these requirements have been developed which provide an equivalent level of protection of the public health and safety. However, since some of these alternatives deviate from the specific wording of Appendix J, exemptions are appropriate for these alternatives. Implicit in the FSAR treatment of the main steam line leakage, as well as the TS requirements for main steam line leakage, are several deviations from the specific requirements of Appendix J. Although PNPP's methods and practices for Appendix J testing have been previously described in correspondence to the NRC, a formal exemption was not recognized to be needed at that time in that the NRC's approval was perceived to be received by the issuance of the PNPP TS. Exemption to four separate paragraphs of 10 CFR 50 Appendix J will document the approvals previously received and incorporated into the TS for main steam line isolation valve testing during the initial licensing of the PNPP. This TS change adds references to footnotes within the TS LCO 3.6.3.1 to clarify which conditions represent exemptions to Appendix J. These exemptions are described in the Bases.

PNPP utilized the criteria described in the Standard Review Plan (SRP), Section 15.6.5, Appendix D, ''Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Main Steam Isolation Valve Leakage Control System (Rev. 1 - July 1981).'' This is an alternative, NRC approved method for assessing the MSIV leakage contribution and determining the radiological consequences.

In accordance with the SRP, the safety analysis for a design basis LOCA includes the maximum main steam line leak rate separately from the maximum containment leak rate. Within Appendix J it is implied that Type A tests are intended to measure the primary containment overall integrated leak rate, but this vas before the SRP Section was developed which allows the MSIV contribution to be accounted for separately in the safety analysis. Therefore, the MSIV leak rate should not be included in the measurement of the ILRT. Including the MSIV leakage in the combined local leak rate limit is also not necessary since a specific Type C MSIV leak rate has been specified in TS 3.6.1.2.

In summary, there is no change in the probability or consequences of any accident since the addition of the references and footnotes to clarify the TS LCO and Actions do not change the design of the plant, nor the operational characteristics of any plant system, nor the procedures by which the Operators run the plant. These changes only cite formal Appendix J exemptions which are requested to document the approval previously received. A formal request for exemption to the applicable paragraphs of 10 CFR 50 Appendix J is also being submitted in a separate letter in conjunction with this proposed TS change.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no design changes being made that would create a new type of accident or malfunction, and the method and manner of plant operation remains unchanged. The only change being made is an exemption to 10 CFR 50 Appendix J which will be cited in the TS to document the implicit and explicit approvals of the PNPP design and testing methods for main steam line isolation valves. The requirements and bases for which the formal exemption is sought are currently presented and implemented in the licensing basis and the TS for PNPP. The objective of the regulation is being met and will continue to be met. The exemption to 10 CFR 50 Appendix J is being submitted in a separate letter in conjunction with this proposed TS change.

3. The proposed changes do not involve a significant reduction in a margin of safety.

These changes do not involve a significant reduction in the margin of safety because they are administrative in nature. The proposed change will only cite the NRC exemption that grants the deviation from Appendix J. The proposed changes do not affect any USAR design bases or accident assumptions. Therefore, the proposed changes do not reduce the margin of safety as defined in the bases for any Technical Specification.

Part B - Revise Surveillance Requirement 4.6.1.2 to Eliminate Unnecessary References and ClarifY the Use of BN-TOP-1

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Surveillance Requirement 4.6.1.2 is proposed to be revised to eliminate the direct reference to the ANSI Standards N45.4 and N56.8 within the text, because these same Standards are listed within Appendix J. It is unnecessary to repeat the references to the Standards within the Technical Specifications because the PNPP is still required to be in compliance with the regulations. No additional benefits are gained and licensee flexibility to upgrade to later versions of the Standards is reduced since a Technical Specification change is necessary to change the version of the Standard to which PNPP is committed. This change removes a redundant requirement to list these Standards in the Technical Specifications. Therefore, this change cannot involve a significant increase in the probability or consequences of an accident because the regulation is still required to be met.

A reference to Topical Report BN-TOP-1 continues to be retained within Surveillance Requirement 4.6.1.2, and the use of the report is clarified to be for test durations less than 24 hours. This reference is retained within the TS since a reference to BN-TOP-1, though not specifically included within Appendix J, is allowed by Section 7.6 of ANSI N45.4-1972 and has been approved for PNPP use by the NRC. The TS Bases are also proposed to be revised to include a statement that the use of BN-TOP-1 is in accordance with Appendix J.

These changes result in no changes to plant systems and have no effect on accident conditions or assumptions. These proposed changes do not affect possible initiating events for accidents previously evaluated, or any system functional requirements. Hence, these changes are purely administrative in that they are designed to eliminate a redundant requirement and clarify the applicability and acceptability of an alternative leak rate testing provision within the TS. These changes do not affect plant operation in any way. Therefore, the proposed changes do not affect the probability or consequences of any accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

There are no design changes being made that would create a new type of accident or malfunction, and the method and manner of plant operation remains unchanged. These changes eliminate a redundant requirement and clarify the applicability and acceptability of alternative leak rate testing provisions within the TS. Since the alternative leak rate testing provisions have been approved by the NRC, the objective of the regulation continues to be met. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

These changes do not involve a significant reduction in the margin of safety because they are administrative in nature and either eliminate a redundant requirement or clarify the applicability and acceptability of an alternative, NRC approved, leak rate testing provision within the TS. The proposed changes do not affect any USAR design bases or accident assumptions. Therefore, the proposed changes do not reduce the margin of safety as defined in the Bases for any Technical Specification.

Part C - Decouple Performance of the Third Type A Test from the Shutdown for the 10-Year Plant Inservice Inspection

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change revises Surveillance Requirement 4.6.1.2.a by removing the second sentence requiring that the third test of each containment Integrated Leak Rate Test (ILRT) set be conducted during the shutdown for the 10-year plant inservice inspection. A request for an exemption to 10 CFR 50 Appendix J, Paragraph III.D.l(a) is also being submitted in conjunction with this proposed change. Note that this change is also included in the proposed Appendix J rule changes currently under consideration and has been approved for several other plants. The deletion of this requirement from the Technical Specifications does not impact plant safety because the 10 CFR 50 Appendix J requirement that three Type A containment ILRT tests to be performed over a 10 year period is not affected. This change only removes an unnecessary connection between the two regulations.

The proposed change results in no changes to plant systems. The proposed change has no effect on accident conditions or assumptions. The proposed change does not affect possible initiating events for accidents previously evaluated, or any system functional requirements. Hence, the proposed change removes an unnecessary tie between regulations and does not affect plant operation in any way.

In summary, there is no change in the probability or consequences of any accident since the revision of the existing Surveillance Requirement to reflect the removal of an unnecessary tie between regulations does not change the design of the plant, nor the operational characteristics of any plant system, nor the procedures by which the Operators run the plant.

2. The propose change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change removes an unnecessary tie between regulations. The objective of the regulation continues to be met. There are no design changes being made that would create a new type of accident or malfunction, and the method and manner of plant operation remains unchanged. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in the margin of safety because they are administrative in nature and remove an unnecessary tie between requirements. The proposed change does not affect any USAR design bases, accident assumptions. or Technical Specification Bases. Therefore, the proposed change does not reduce the margin of safety as defined in the bases for any TS.

Based upon the above considerations, it has been concluded that the proposed changes do not involve significant hazards considerations.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Gail H. Marcus

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: June 9 and 30, 1995

Description of amendment request: The licensee has requested a one-time extension of the performance intervals for certain Technical Specification Surveillance Requirements (SRs). Affected SRs include valve testing, and undervoltage instrumentation testing.

Basis for proposed no significant hazards considerationdetermination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS change requests one-time only extensions of the surveillance intervals related to: a) ASME Section XI valve leak rate, stroke and timing, and position indication testing; b) Accident Monitoring Instrumentation related to valve position indication testing; c) Division 1, 2, and 3 Degraded Voltage and Undervoltage instrumentation LSFT; and, d) leak rate testing for hydrostatically tested containment isolation valves.

Based on the discussion in the License Amendment Request which shows:

i) The extension of the interval for ASME Section XI stroke and timing, leak rate measurement and position indication testing requirements are acceptable based on results of past testing which indicates a margin to TS limits will be maintained;

ii) The extension of the interval for Position Indication Calibration as specified in Table 4.3.7.5-1, Item 17 is acceptable based on the testing results from the past two refueling outages that indicate no failures have occurred:

iii) LSFT interval extension for the Division 1, 2, and 3 Degraded Voltage and Undervoltage instrumentation is acceptable based on the NRC Safety Evaluation Report (Peach Bottom Atomic Power Plant, Units 2 and 3, dated August 2, 1993) which supported extension of the interval for LSFT from 18 to 24 months. This was based on the small probability of relay or contact failure relative to mechanical component failure probability and, therefore, the increase in LSFT interval represented no significant change in the overall safety system unavailability; and,

iv) The extension of the interval for hydrostatic leak testing of containment isolation valves is acceptable based on the consistently low past leak rate data which is a small percentage of the TS limits.

Therefore, from the above it is shown that the proposed changes will not significantly increase the probability of an accident previously evaluated.

The proposed TS change requests one-time only extensions of the surveillance intervals related to TS SR 4.3.3.1, Table 4.3.3.1-1, Items D.1 and D.2, Division 1, 2, and 3 Degraded Voltage and Undervoltage instrumentation calibration. [...] extension of the interval for this instrumentation is acceptable based on the testing results from the past two refueling outages. No failures have occurred which would negate the assurance that the instrumentation would function as required for the requested extended period. Accordingly, the proposed change will not significantly increase the probability or consequences of an accident previously evaluated.

2. The proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS change requests one-time extensions of the surveillance intervals for ASME Section XI valve testing, instrumentation calibration, instrument channel LSFT, containment isolation valve hydrostatic leak rate testing. The proposed changes do not necessitate a physical alteration to the plant (no new or different type of equipment will be installed). In that the requested extension durations are small as compared to the overall interval allowed by TS, NRC and industry evaluations support extension of LSFT, and past testing results provide confidence of no effect on equipment availability by extending the surveillance interval, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS change requests one-time extensions of the surveillance intervals for the Division 1, 2, and 3 Undervoltage and Degraded Voltage instrumentation calibration. The proposed changes do not necessitate a physical alteration to the plant (no new or different type of equipment will be installed). In that the requested extension durations are small as compared to the overall interval allowed by TS and past testing results provide confidence of no effect on equipment availability by extending the surveillance interval, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change will not involve a significant reduction in the margin of safety.

The proposed TS change requests a one-time extension of the surveillance intervals for ASME Section XI valve testing, instrumentation calibration, instrument channel LSFT, and containment isolation valve hydrostatic leak rate testing. The proposed changes do not necessitate a physical alteration to the plant (no new or different type of equipment will be installed). In that the requested extension durations are small as compared to the overall interval allowed by TS, NRC and industry evaluations support extension of LSFT, and past testing results provide confidence of no effect on equipment availability by extending the surveillance interval, the change does not involve a significant reduction in the margin of safety.

The proposed TS change requests a one-time extension of the surveillance intervals for the division 1, 2, and 3 Undervoltage and Degraded Voltage instrumentation calibration. The proposed changes do not necessitate a physical alteration to the plant (no new or different type of equipment will be installed). In that the requested extension durations are small as compared to the overall interval allowed by TS and past testing results provide confidence of no effect on equipment availability by extending the surveillance interval, the change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room Location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Gail H. Marcus

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: July 19, 1995

Description of amendments request: Amend the Sequoyah Nuclear Plant, Units 1 and 2 Technical Specification to incorporate new requirements associated with steam generator tube inspections and repair.

Date of publication of individual notice in the **Federal Register:** August 1, 1995 (60 FR 39198)

Expiration date of individual notice: August 31, 1995

Local Public Document Room Location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Notice Of Issuance Of Amendments To Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of applications for amendments: December 30, 1993 and July 12, 1994. The December 30, 1993, application was supplemented by letters dated November 30, 1994, May 24, 1995, and June 21, 1995, and the July 12, 1994, application was supplemented by letter dated June 21, 1995.

Brief description of amendments: The amendments (1) revise the degraded voltage relay trip setpoint and (2) enhance the current presentation of the information regarding the loss-of-voltage relay setpoint. A time-voltage curve has been added to the technical specifications as a more accurate characterization of the inverse-time relay response.

Date of issuance: July 21, 1995

Effective date: July 21, 1995, to be implemented within 45 days of issuance.

Amendment Nos.: Unit 1 - Amendment No. 96; Unit 2 - Amendment No. 84; Unit 3 - Amendment No. 67

Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 8, 1994 and August 17, 1994 (59 FR 29625 and 59 FR 42334) The November 30, 1994, May 24, 1995, and June 21, 1995, letters provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004.

Carolina Power & Light Company, et al., Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: February 6, 1995

Brief description of amendment: The amendment allows the relocation of cycle-specific core operating limits of Figure 3.1-1, Shutdown Margin versus Boron Concentration in Technical Specification (TS) 3.1.1.2, Shutdown Margin- Modes 3, 4, and 5, to the plant Core Operating Limits Report.

Date of issuance: August 1, 1995

Effective date: August 1, 1995

Amendment No. 59

Facility Operating License No. NPF-63. Amendment revises the Technical Specifications

Date of initial notice in **Federal Register:** March 15, 1995 (60 FR 14017) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Carolina Power & Light Company, et al., Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: March 30, 1995, as supplemented July 6, 1995. The July 6, 1995, submittal did not change the initial no significant hazards consideration determination; it contained clarifying information only.

Brief description of amendment: The amendment revises the Emergency Diesel Generator (EDG) surveillance requirements contained in TS 3/48.1.1.2 to be consistent with NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," and to eliminate the need for duplicate EDG testing being performed to satisfy the requirements of the Station Blackout Rule and the Maintenance Rule.

Date of issuance: August 1, 1995

Effective date: August 1, 1995

Amendment No.: 60

Facility Operating License No. NPF-63. Amendment revises the Technical Specifications

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20515) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Commonwealth Edison Company, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: June 8, 1995, which superseded the December 16, 1994, request in its entirety, and additional correspondence dated November 30, 1994, April 27, May 5, May 11 and June 23, 1995.

Brief description of amendments: The amendments revised Figure 3.4-4a "Nominal PORV Pressure Relief Setpoint Versus RCS Temperature for the Cold Overpressure Protection (LTOP) System" in the Braidwood Unit 1's Technical Specifications. The revision extends the applicability of Figure 3.4-4a from 5.37 effective full power years (EFPY) to 16 EFPY. In addition, the amendments remove the 638 psig administrative limit line from the LTOPS curve, because the appropriate instrument uncertainties and discharge piping pressure limits have been incorporated in the new curve. Finally, the amendments contains administrative changes to Figure 3.4-4a and its associated index page.

Date of issuance: July 24, 1995

Effective date: July 24, 1995

Amendment Nos.: 64 and 64

Facility Operating License Nos. NPF-72 and NPF-77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32360). The June 23, 1995, letter, corrected a collating error in the June 8, 1995, submittal and did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: March 23, 1994, as supplemented on July 26, 1994, and subsequently superseded by a submittal dated

February 15, 1995. The February 15, 1995, request was supplemented on February 28, 1995.

Brief description of amendments: The amendments approve a maximum moderator temperature coefficient (MTC) of +7 pcm/°F and relocate specification of the cycle specific MTC from the Technical Specifications to the operating limits report. The staff also approved the methodology proposed by the licensee for ensuring that the plants continue to meet the anticipated transient without scram (ATWS) rule (10 CFR 50.62) during operation with cycle specific MTCs.

Date of issuance: July 27, 1995Effective date: Immediately, to be implemented within 30 days.

Amendment Nos.: Byron Units 1 and 2 - 73, 73 and Braidwood Units 1 and 2 - 65, 65

Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 12, 1995 (60 FR 18623) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 27, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: July 29, 1992, as supplemented January 14, 1993, February 16, 1993, and May 9, 1995.

Brief description of amendments: The amendments upgrade the current custom Technical Specifications for Dresden and Quad Cities to the Standard Technical Specifications contained in NUREG-0123, ''Standard Technical Specification General Electric Plants BWR/4.'' These amendments upgrade only Section 3/4.3, ''Reactivity Control.''

Date of issuance: July 27, 1995 Effective date: Immediately, to be implemented no later than December 31, 1995, for Dresden Station and June 30, 1996, for Quad Cities Station.

Amendment Nos.: 137, 131, 158, and 154

Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 23, 1993 (58 FR 34071) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated June 27, 1995. No significant hazards consideration comments received: No

Local Public Document Room Location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of application for amendments: December 14, 1994

Brief description of amendments: The amendments revise the surveillance test intervals and allowed outage times for certain actuation instrumentation in the reactor protection, isolation, emergency core cooling, control rod withdrawal block, monitoring and feedwater/main turbine trip systems. The amendments also include changes to the feedwater/ main turbine trip limiting condition for operation required actions, several mode related changes to the nuclear instrumentation and rod block specifications, shiftly channel check requirements for several systems, and several editorial changes to correct errors and remove outdated footnotes.

Date of issuance: August 2, 1995

Effective date: Immediately, to be implemented within 90 days.

Amendment Nos.: 104 and 90

Facility Operating License Nos. NPF-11 and NPF-18: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11128) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 2, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Commonwealth Edison Company, Docket No. 50-295, Zion Nuclear Power Station, Unit 1, Lake County, Illinois

Date of application for amendment: May 17, 1995, as supplemented on June 2, June 16, and July 12, 1995.

Brief description of amendment: The amendment allows a limited number of steam generator tubes with roll transition indications to remain in service until the September 1995 refueling outage.

Date of issuance: July 26, 1995

Effective date: July 26, 1995

Amendment No.: 167

Facility Operating License No. DPR-39: The amendment revises the Technical Specifications. The June 2, June 16, and July 12, 1995, submittals provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The information, however, included changes to details of the administrative limits mentioned in the initial proposed no significant hazards consideration determination.Public comments requested as to proposed no significant hazards consideration determination: Yes (60 FR 27798). This notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by June 26, 1995, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendments, finding of exigent circumstances and final no significant hazards consideration determination is contained in a Safety Evaluation dated July 26, 1995.

Local Public Document Room Location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

Consumers Power Company, Docket No. 50-155, Big Rock Point Plant, Charlevoix County, Michigan

Date of application for amendment: December 15, 1994

Brief description of amendment: The amendment revises Technical Specification 11.3.1.5 ACTION a. to eliminate the need to demonstrate that the actuation circuitry of the unaffected reactor depressurization system channels is operable. In addition, the amendment makes an editorial change to correct a typographical error.

Date of issuance: July 28, 1995

Effective date: July 28, 1995

Amendment No.: 115

Facility Operating License No. DPR-6. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** April 26, 1995 (60 FR 20516) The Commission's related evaluation of the amendment is contained in a Safety

Evaluation dated July 28, 1995. No significant hazards consideration comments received: No.

Local Public Document Room Location: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

Consumers Power Company, Docket No. 50-155, Big Rock Point Plant, Charlevoix County, Michigan

Date of application for amendment: March 4, 1993, as revised April 14, 1993, as supplemented April 19 and May 31, 1995

Brief description of amendment: The amendment revises the Technical Specifications (TS) to conform to the wording of the revised 10 CFR Part 20, ''Standards for Protection Against Radiation,'' and to reflect a separation of chemistry and radiation protection responsibilities.

Date of issuance: August 2, 1995

Effective date: August 2, 1995

Amendment No.: 16

Facility Operating License No. DPR-6. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 12, 1993 (58 FR 28053), as corrected June 1, 1993 (58 FR 31222). The supplemental submittals were noticed on June 21, 1995 (60 FR 32361). The Commission's related evaluation of the amendment is contained in a Safety Evaluation datedNo significant hazards consideration comments received: No.

Local Public Document Room Location: North Central Michigan College, 1515 Howard Street, Petoskey, Michigan 49770.

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: April 7, 1994, as supplementedApril 27, 1995.

Brief description of amendment: This amendment relocates certain Technical Specifications (TS) that contain fuel cycle-specific parameter limits that change with core reloads to a Core Operating Limits Report. TS bases have also been revised to refer to limits relocated to the COLR. A portion of the amendment request was denied. A separate Notice of Denial of Amendment has been sent to the **Federal Register** for publication.

Date of issuance: July 26, 1995

Effective date: July 26, 1995

Amendment No.: 169

Facility Operating License No. DPR-20. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 25, 1994 (59 FR 27053) The April 27, 1995, submittal provided clarifying information which was within the scope of the initial application and did not affect the staff's initial proposed no significant hazards consideration findings. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 26, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: April 12, 1995

Brief description of amendments: The amendments delete Technical Specification (TS) 3/4.3.4, ''Turbine Overspeed Protection,'' and its associated Bases. The deletion of TS 3/4.3.4 and its Bases provides Duke Power Company the flexibility to implement the manufacturer's recommendations for turbine steam valve surveillance test requirements. These test requirements will be contained in the Selected Licensee Commitment Manual.

Date of issuance: July 21, 1995

Effective date: As of the date of issuance to be implemented within 30 days from the date of issuance

Amendment Nos.: 131 and 125

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32361) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 1995. No significant hazards consideration comments received: No

Local Public Document Room Location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: January 18, 1995.

Brief description of amendments: The amendments relocate the requirements for the seismic instrumentation, meteorological instrumentation, and loose-part detection system, and the associated Bases and surveillance requirements, from the TS to the Selected Licensee Commitment Manual (Chapter 16 of the FSAR). This will allow future changes to these controls to be performed under the provisions of 10 CFR 50.59. No changes are being made to the technical content of the affected TS pages.

Date of issuance: July 24, 1995

Effective date: As of the date of issuance to be implemented within 30 days

Amendment Nos.: 132 and 126

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24910) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: April 12, 1995

Brief description of amendments: The amendments delete Technical Specification (TS) 3/4.3.4, ''Turbine Overspeed Protection,'' and its associated Bases. The deletion of TS 3/4.3.4 and its associated Bases provides Duke Power Company the flexibility to implement the manufacturer's recommendations for turbine steam valve surveillance test requirements. These test requirements will be contained in the Selected Licensee Commitment (SLC) Manual. The SLC Manual is Chapter 16 of the Updated Final Safety Analysis Report.

Date of issuance: August 2, 1995

Effective date: As of the date of issuance to be implemented within 30 days from the date of issuance

Amendment Nos.: 156 and 138

Facility Operating License Nos. NPF-9 and NPF-17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32362) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 2, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: September 28, 1994, as supplemented by letters dated May 3 and June 14, 1995.

Brief description of amendments: The amendments revise Technical Specification Tables 3.3-3, 3.3-4, 3.3-5, and 4.3-2 of the Engineered Safety Features Actuation System Instrumentation tables to update the ''Loss of Power'' function.

Date of issuance: August 2, 1995

Effective date: As of the date of issuance to be implemented within 60 days, or 60 days after the completion date of the Unit 2 modification, whichever is later.

Amendment Nos.: 157 and 139

Facility Operating License Nos. NPF-9 and NPF-17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65811) The May 3 and June 14, 1995, letters provided clarifying information that did not change the scope of the September 28, 1994, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 2, 1995. No significant hazards consideration comments received: No.

Local Public Document Room Location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: January 18, 1995

Brief description of amendments: The amendments delete selected Technical Specification (TS) requirements related to instrumentation from the TS, and relocate them to the Selected Licensee Commitment (SLC) Manual, with their associated Bases and surveillance requirements. No changes are being made to the technical content of the affected TS pages. Future changes to the SLC Manual (Chapter 16 of the Final Safety Analysis Report) will be controlled by the provisions of 10 CFR 50.59. The relocated requirements include the following:

TS 3/4.3.3.3, Seismic Instrumentation

TS 3/4.3.3.4, Meteorological Instrumentation

TS 3/4.10, Loose-Part Detection System

Date of issuance: August 2, 1995

Effective date: As of the date of issuance to be implemented within 30 days from the date of issuance

Amendment Nos.: 158 and 140

Facility Operating License Nos. NPF-9 and NPF-17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11132) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 2, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: February 4, 1994, as supplemented June 29, 1995.

Brief description of amendments: These amendments modify the Technical Specifications (TSs) related to containment air locks (TSs 1.8, 3/4.6.1.1 and 3/4.6.1.3) and associated Bases to make them as close to the NRC's Improved Standard Technical Specifications (NUREG-1431) as the plant-specific design will permit. The changes in TS 3/4.6.1.1 and 3/4.6.1.3 modify surveillance requirements and limiting conditions for operation and effect numerous administrative and format changes.

Date of issuance: July 26, 1995

Effective date: Units 1 and 2, as of the date of issuance and shall be implemented within 60 days.

Amendment Nos.: 190 and 72

Facility Operating License Nos. DPR-66 and NPF-73: Amendments revised the Units 1 and 2 Technical Specifications, and the Unit 2 License.

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37070) The June 29, 1995 letter did not change the original no significant hazards consideration determination or expand the scope of the July 20, 1994 Federal Register notice.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 26, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: May 12, 1995

Brief description of amendment: The amendment removed the specific scheduling requirements for Type A containment leakage rate tests from the Technical Specifications for Waterford 3 and replaced these requirements with a requirement to perform Type A, testing in accordance with Appendix J to 10 CFR Part 50. The proposed changes adopt the wording for primary containment integrated leak rate testing that is consistent with the requirements of the Combustion Engineering Improved Standard Technical Specifications (NUREG 1432). The proposed changes also include several administrative changes.

Date of issuance: August 3, 1995

Effective date: August 3, 1995, to be implemented within 60 days of issuance.

Amendment No.: 110

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29876) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 3, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: October 13, 1994, as supplemented by letters dated January 13 and May 4, 1995.

Brief description of amendments: The amendments revise the Technical Specifications to lower the anticipated transient without scram-recirculation pump trip (ATWS-RPT) setpoint by approximately 2 feet 2 inches to minimize the potential for RPTs following reactor scram, and allow restarting the recirculation pump following an RPT when the temperature differential between the coolant at the reactor bottom head and the reactor steam dome cannot be obtained, provided certain conditions are met.

Date of issuance: July 21, 1995

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment Nos.: 196 and 136

Facility Operating License Nos. DPR-57 and NPF-5. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR

65813). The January 13 and May 4, 1995, letters provided clarifying information that did not change the scope of the October 13, 1994, application and initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 21, 1995. No significant hazards consideration comments received: No

Local Public Document Room Location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513.

GPU Nuclear Corporation, et al., Docket No. 50-289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of application for amendment: June 1, 1995

Brief description of amendment: The amendment revises the TMI-1 Technical Specifications to allow the use of two zirconium-based advanced fuel rod cladding materials manufactured by the Babcock & Wilcox Fuel Company.

Date of issuance: July 24, 1995

Date of issuance: July 24, 1995

Effective date: July 24, 1995

Amendment No.: 194

Facility Operating License No. DPR-50. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32366) The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated July 24, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: Law/Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: May 13, 1993 as supplemented by letter dated January 31, 1995

Brief description of amendment: The amendment revises Attachment 3 of the license conditions to remove several license conditions pertaining to the Division I and II Transamerica Delaval, Inc. emergency diesel generators. The conditions pertain to engine overhaul frequency, maintenance and surveillance program, and inspection of crankshafts, cylinder heads, engine block, and turbochargers.

Date of issuance: July 25, 1995

Effective date: July 25, 1995

Amendment No.: 82

Facility Operating License No. NPF-47. The amendment revised the operating license.

Date of initial notice in **Federal Register:** August 4, 1993 (58 FR 41505) The additional information contained in the supplemental letter dated January 31, 1995, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 25, 1995.No significant hazards consideration comments received. No.

Local Public Document Room Location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: April 27, 1995, as supplemented by letters dated May 4 and 25, 1995.

Brief description of amendments: The amendments revised the tables associated with Technical Specifications (TSs) 3/4.3.3.5, Remote Shutdown System, to eliminate the requirement for core exit thermocouples (CETs). The amendments also revised the tables associated with TS 3/4.3.3.6, Accident Monitoring Instrumentation, to require two operable channels of CETs, where each channel is required to have at least two operable CETs per core quadrant. Each channel is also required to have at least four operable CETs in at least one quadrant to support the operability of the subcooling margin monitors.

Date of issuance: July 24, 1995

Effective date: July 24, 1995

Amendment Nos.: Unit 1 - Amendment No. 77; Unit 2 - Amendment No. 66

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32366) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 24, 1995. No significant hazards consideration comments received: No

Local Public Document Room Location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 2, 1995

Brief description of amendments: The amendments revised Technical Specifications 3.4.2.2. and 3.7.1.1 (Table 3.7-2) by relaxing the lift setting tolerances of the pressurizer safety valves from plus or minus 1% to plus or minus 2% and the main steam safety valves from plus or minus 1% to plus or minus 3%, respectively. In addition, a footnote was added to require that the pressurizer safety valves and main steam safety valves setpoint tolerances be restored to within plus or minus 1% whenever a lift setting is determined to be outside plus or minus 1% following valve testing.

Date of issuance: July 25, 1995

Effective date: July 25, 1995, to be implemented within 30 days of issuance.

Amendment Nos.: Unit 1 - Amendment No. 78; Unit 2 - Amendment No. 67

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29877) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 25, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: March 7, 1995, as supplemented on June 7, 1995.

Brief description of amendment: The amendment adds an Exception to Technical Specifications 3.6.A and 3.6.C. The Exception permits reduced component cooling water flow for short periods of time, while component cooling water heat exchangers are shifted.

Date of issuance: July 24, 1995

Effective date: As of the date of issuance, to be implemented within 30 days.

Amendment No.: 151

Facility Operating License No. DPR-36: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24911) The June 7, 1995, submittal provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

Northeast Nuclear Energy Company, Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: May 24, 1995

Brief description of amendment: The amendment permits an individual who does not have a current senior reactor operator (SRO) license for Millstone Unit 1 to hold the Operations Manager position. In this case, the Operations Manager position would require the individual to have previously held an SRO license at a boiling water reactor and the individual serving in the capacity of the Assistant Operations Manager to hold a current SRO license for Millstone Unit 1. In addition, the amendment renumbers the applicable sections.

Date of issuance: July 24, 1995

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 83

Facility Operating License No. DPR-21. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 21, 1995 (60 FR 32370) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Northeast Nuclear Energy Company, Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: April 18, 1995

Brief description of amendment: The amendment allows the use of the ANSI/ANS 5.1-1979 decay heat model for the post-loss of coolant accident containment cooling analysis.

Date of issuance: July 24, 1995

Effective date: As of the date of issuance to be implemented immediately.

Amendment No.: 84

Facility Operating License No. DPR-21. Amendment revised the license.

Date of initial notice in **Federal Register:** May 10, 1995 (60 FR 24911). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 24, 1995. No significant hazards consideration comments received: No.

Local Public Document Room Location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: April 28, 1995

Brief description of amendment: The amendment revises the diesel generator fuel oil testing that is performed on new fuel prior to the addition of new fuel to the storage tank.

Date of issuance: July 26, 1995

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 118

Facility Operating License No. NPF-49. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 6, 1995 (60 FR 29881) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 26, 1995. No significant hazards consideration comments received: No.

Local Public Document Room Location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station,Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: November 14, 1994 as supplemented by letter dated April 10, 1995.

Brief description of amendments: These amendments relocate Nuclear Review Board (NRB) review requirements, Independent Safety Engineering Group (ISEG) requirements, and certain review and audit requirements from the TS to the Peach Bottom Quality Assurance Program.

Date of issuance: July 25, 1995

Effective date: July 25, 1995

Amendments Nos.: 208 and 212

Facility Operating License Nos. DPR-44 and DPR-56: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65822) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 25, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: July 27, 1994, as supplemented May 26, July 10, and July 25, 1995

Brief description of amendment: This amendment revises the Allowed Out-of-Service Times (AOTs) for Inoperable Station Service Water System (SSWS) pumps, inoperable safety Auxiliaries Cooling System (SACS) pumps, and inoperable Emergency Diesel Generators (EDGs). In addition, this amendment also allows on-line maintenance of the EDGs.

Date of issuance: August 1, 1995

Effective date: August 1, 1995

Amendment No.: 75

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 31, 1994 (59 FR 45033) The supplemental letters did not change the original no significant hazards consideration determination nor the original Federal Register notice. The

Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: April 25, 1994, as supplemented July 24, 1995

Brief description of amendment: This amendment eliminates the requirement from the Hope Creek Technical Specifications to perform Type C leak rate tests, in accordance with 10 CFR Part 50, Appendix J, of identified containment isolation valves that penetrate the primary containment and terminate below the minimum water level in the suppression chamber (torus). The valves are still subject to testing in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

Date of issuance: August 1, 1995

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 76

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** June 8, 1994 (59 FR 29632) The supplemental letter did not change the original no significant hazards consideration determination nor the original Federal Register notice.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070.

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: April 18, 1995

Brief description of amendments: The amendments delete the quarterly leak rate test for the containment pressure-vacuum relief valves that is currently required because of the valves' resilient seat material. The changes are being made to accommodate replacement of the resilient valve seat material with a hard seat (metal-to-metal) design. The valves would remain in the 10 CFR Part 50, Appendix J, Type C leak rate test program.

Date of issuance: August 1, 1995

Effective date: Unit 1, As of the date of issuance, to be implemented prior to restart following the twelfth refueling outage; Unit 2, As of the date of issuance, to be implemented prior to restart following the current refueling outage.

Amendment Nos.: 172 and 153

Facility Operating License Nos. DPR-70 and DPR-75. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 23, 1995 (60 FR 27342) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of application for amendments: August 26, 1994

Brief description of amendments: These amendments revise Technical Specification 3/4.7.5, ''Control Room Emergency Air Cleanup System,'' to provide an exception to Limiting Condition for Operation 3.0.4 for Modes 5 and 6 and for a defueled configuration. These amendments also add the applicability statement ''or during movement of irradiated fuel assemblies.''

Date of issuance: July 26, 1995

Effective date: July 26, 1995

Amendment Nos.: Unit 2 - Amendment No. 123; Unit 3 - Amendment No. 112

Facility Operating License Nos. NPF-10 and NPF-15: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55891) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated July 26, 1995.No significant hazards consideration comments received: No.

Local Public Document Room Location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713.

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: December 16, 1994; supplemented July 19, 1995 (TS 94-06)

Brief description of amendments: The amendments replace the present Auxiliary Feedwater system Specification 3/4.7.1.2 with new specifications that are modeled after the Westinghouse Standard Technical Specifications.

Date of issuance: August 2, 1995

Effective date: August 2, 1995

Amendment Nos.: 206 and 196

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.

Date of initial notice in **Federal Register:** February 1, 1995 (60 FR 6309) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 2, 1995.No significant hazards consideration comments received: None

Local Public Document Room Location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of application for amendments: November 29, 1994

Brief description of amendments: These amendments allow the use of ZIRLO, a new zirconium-based alloy, as a fuel cladding material.

Date of issuance: July 27, 1995

Effective date: July 27, 1995

Amendment Nos.: 202 and 202

Facility Operating License Nos. DPR-32 and DPR-37: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 508) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated July 27, 1995.No significant hazards consideration comments received: No

Local Public Document Room Location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Notice Of Issuance Of Amendments to facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

During the period since publication of the last biweekly notice, the

Commission has issued the following amendments. The Commission has determined for each of these amendments that the application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By September 15, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of application for amendment: July 28, 1995

Brief description of amendment: This amendment deletes the portion of License Condition 2.C.(1) that references Attachment 1. Attachment 1 requires the pump in the keepwarm system on the emergency diesel generator to satisfy the requirements of the American Society of Mechanical Engineers Code, Section III, Class 3.

Date of issuance: August 3, 1995I11*Effective date:* August 3, 1995

Amendment No.: 88

Facility Operating License No. NPF-42: The amendment revised the operating license.Public comments requested as to proposed no significant hazards consideration: No.The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated August 3, 1995.

Local Public Document Room Location: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

Dated at Rockville, Maryland, this 16th day of August 1995.

For The Nuclear Regulatory Commission

Jack W. Roe,

Director, Division of Reactor Projects - III/IV Office of Nuclear Reactor Regulation

[Doc. 95–20122 Filed 8–15–95; 8:45 am]

BILLING CODE 7590–01–F

Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement

AGENCY: Nuclear Regulatory Commission.

ACTION: Final policy statement.

SUMMARY: This statement presents the policy that the Nuclear Regulatory Commission (NRC) will follow in the use of probabilistic risk assessment (PRA) methods in nuclear regulatory matters. The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach. The pertinent comments received from the published draft policy statement are reflected in this final policy statement. This policy statement will be implemented through the execution of the NRC's PRA Implementation Plan.

EFFECTIVE DATE: August 16, 1995.

ADDRESSES: The proposed policy statement and the comments received may be examined at: NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Anthony Hsia, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 415–1075.

SUPPLEMENTARY INFORMATION:

I. Background

The NRC has generally regulated the use of nuclear material based on deterministic approaches. Deterministic approaches to regulation consider a set of challenges to safety and determine how those challenges should be mitigated. A probabilistic approach to regulation enhances and extends this traditional, deterministic approach, by: (1) Allowing consideration of a broader set of potential challenges to safety, (2) providing a logical means for prioritizing these challenges based on risk significance, and (3) allowing consideration of a broader set of resources to defend against these challenges.

Until the accident at Three Mile Island (TMI) in 1979, the Atomic Energy Commission (now the NRC), only used probabilistic criteria in certain specialized areas of licensing reviews. For example, human-made hazards (e.g., nearby hazardous materials and aircraft) and natural hazards (e.g., tornadoes, floods, and earthquakes) were typically addressed in terms of probabilistic arguments and initiating frequencies to assess site suitability. The Standard Review Plan (NUREG–0800) for licensing reactors and some of the Regulatory Guides supporting NUREG–0800 provided review and evaluation guidance with respect to these probabilistic considerations.

The TMI accident substantially changed the character of the analysis of severe accidents worldwide. It led to a substantial research program on severe accident phenomenology. In addition,